

**Millstone Power Station Unit 3
Safety Analysis Report**

Chapter 19: License Renewal

CHAPTER 19—LICENSE RENEWAL

Table of Contents

<u>Section</u>	<u>Title</u>	<u>Page</u>
19.0	INTRODUCTION	19.0-1
19.1	AGING MANAGEMENT.....	19.1-1
19.1.1	Aging Management Programs	19.1-1
19.1.2	Time Limited Aging Analyses Aging Management Programs:	19.1-2
19.1.3	References for Section 19.1	19.1-2
19.2	PROGRAMS THAT MANAGE THE EFFECTS OF AGING ON STRUCTURES AND COMPONENTS WITHIN THE SCOPE OF LICENSE RENEWAL	19.2-1
19.2.1	Aging Management Programs	19.2-1
19.2.1.1	Battery Rack Inspections	19.2-1
19.2.1.2	Boric Acid Corrosion.....	19.2-2
19.2.1.3	Buried Pipe Inspection Program	19.2-2
19.2.1.4	Chemistry Control for Primary Systems Program.....	19.2-3
19.2.1.5	Chemistry Control for Secondary Systems Program.....	19.2-4
19.2.1.6	Closed-Cycle Cooling Water System	19.2-4
19.2.1.7	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	19.2-5
19.2.1.8	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits.....	19.2-6
19.2.1.9	Fire Protection Program.....	19.2-7
19.2.1.10	Flow-Accelerated Corrosion.....	19.2-8
19.2.1.11	Fuel Oil Chemistry.....	19.2-8
19.2.1.12	General Condition Monitoring.....	19.2-9
19.2.1.13	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	19.2-10
19.2.1.14	Infrequently Accessed Areas Inspection Program.....	19.2-11
19.2.1.15	Inservice Inspection Program: Containment Inspections	19.2-12
19.2.1.16	Inservice Inspection Program: Reactor Vessel Internals	19.2-12
19.2.1.17	Inservice Inspection Program: Systems, Components and Supports.....	19.2-14

CHAPTER 19—LICENSE RENEWAL
Table of Contents (Continued)

<u>Section</u>	<u>Title</u>	<u>Page</u>
19.2.1.18	Inspection Activities: Load Handling Cranes and Devices	19.2-15
19.2.1.19	Reactor Vessel Surveillance	19.2-16
19.2.1.20	Service Water System (Open-Cycle Cooling)	19.2-16
19.2.1.21	Steam Generator Structural Integrity	19.2-17
19.2.1.22	Structures Monitoring Program	19.2-18
19.2.1.23	Tank Inspection Program	19.2-19
19.2.1.24	Work Control Process	19.2-20
19.2.1.25	Bolting Integrity Program	19.2-22
19.2.2	References for Section 19.2	19.2-22
19.3	TIME-LIMITED AGING ANALYSIS	19.3-1
19.3.1	Reactor Vessel Neutron Embrittlement	19.3-1
19.3.1.1	Upper Shelf Energy	19.3-1
19.3.1.2	Pressurized Thermal Shock	19.3-1
19.3.1.3	Pressure-Temperature Limits	19.3-2
19.3.2	Metal Fatigue	19.3-2
19.3.2.1	Millstone Unit 3 Class 1 Components	19.3-3
19.3.2.2	Non-Class 1 Components	19.3-4
19.3.2.3	Environmentally Assisted Fatigue	19.3-4
19.3.3	Environmental Qualification (EQ) of Electric Equipment	19.3-6
19.3.4	Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis	19.3-6
19.3.4.1	Containment Liner Plate	19.3-6
19.3.4.2	Containment Penetrations	19.3-6
19.3.5	Other Plant-Specific Time-Limited Aging Analyses	19.3-7
19.3.5.1	Crane Load Cycle Limit	19.3-7
19.3.5.2	Reactor Coolant Pump Flywheel	19.3-7
19.3.5.3	Leak-Before-Break	19.3-8
19.3.5.4	Containment Subfoundation	19.3-9
19.3.6	References for Section 19.3	19.3-9

CHAPTER 19—LICENSE RENEWAL
Table of Contents (Continued)

<u>Section</u>	<u>Title</u>	<u>Page</u>
19.4	TLAA SUPPORT PROGRAMS	19.4-1
19.4.1	Electrical Equipment Qualification	19.4-1
19.4.2	Metal Fatigue of Reactor Coolant Pressure Boundary	19.4-1
19.5	EXEMPTIONS	19.5-1
19.6	LICENSE RENEWAL COMMITMENTS	19.6-1
19.6.1	References for Section 19.6	19.6-1

CHAPTER 19—LICENSE RENEWAL

List of Tables

<u>Number</u>	<u>Title</u>
19.6 - 1	License Renewal Commitments

CHAPTER 19 - LICENSE RENEWAL

19.0 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix, which includes the following sections, comprises the FSAR supplement:

- Section 19.1 contains a listing of the aging management programs and the status of the program at the time the License Renewal Application was submitted.
- Section 19.2 contains a description of the programs for managing the effects of aging.
- Section 19.3 contains the evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation.
- Section 19.4 contains a summarized description of the programs that support the TLAAs.
- Section 19.5 contains a summarized description of the plant-specific exemptions.
- Section 19.6 contains a matrix of the license renewal commitments.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the unit's previous 40 year operating license expiration date. Unless otherwise identified, references to the Operating License are considered a reference to the Renewed Operating License.

19.1 AGING MANAGEMENT

19.1.1 AGING MANAGEMENT PROGRAMS

The aging management programs for Millstone Unit3 are described in the following sections. The programs are either consistent with generally accepted industry methods as discussed in NUREG-1801 (Reference 19.1-1), require enhancements to be consistent with generally accepted industry standards, or are site-specific programs.

The following list reflects the status of these programs at the time this section was included in the FSAR and provides a historical perspective of their status at the completion of the NRC review of the License Renewal Application. The implementation status of the listed programs will change as new programs are developed and enhancements to existing programs are completed. Commitments for program additions and enhancements are identified in the appropriate sections.

1. Battery Rack Inspections (Section 19.2.1.1) (Existing - Requires Enhancement).
2. Boric Acid Corrosion (Section 19.2.1.2) (Existing).
3. Buried Pipe Inspection Program (Section 19.2.1.3) (Existing - Requires Enhancement).
4. Chemistry Control for Primary Systems Program (Section 19.2.1.4) (Existing).
5. Chemistry Control for Secondary Systems Program (Section 19.2.1.5) (Existing).
6. Closed-Cycle Cooling Water System (Section 19.2.1.6) (Existing - Requires Enhancement.)
7. Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section 19.2.1.7) (To Be Developed).
8. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section 19.2.1.8) (Existing - Requires Enhancement).
9. Fire Protection Program (Section 19.2.1.9) (Existing - Requires Enhancement).
10. Flow-Accelerated Corrosion (Section 19.2.1.10) (Existing).
11. Fuel Oil Chemistry (Section 19.2.1.11) (Existing).
12. General Condition Monitoring (Section 19.2.1.12) (Existing - Requires Enhancement).

13. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section 19.2.1.13) (Existing - Requires Enhancement).
14. Infrequently Accessed Areas Inspection Program (Section 19.2.1.14) (To Be Developed).
15. Inservice Inspection Program: Containment Inspections (Section 19.2.1.15) (Existing).
16. Inservice Inspection Program: Reactor Vessel Internals (Section 19.2.1.16) (Existing - Requires Enhancement).
17. Inservice Inspection Program: Systems, Components and Supports (Section 19.2.1.17) (Existing - Requires Enhancement).
18. Inspection Activities: Load Handling Cranes and Devices (Section 19.2.1.18) (Existing - Requires Enhancement).
19. Reactor Vessel Surveillance (Section 19.2.1.19) (Existing).
20. Service Water System (Open-Cycle Cooling) (Section 19.2.1.20) (Existing).
21. Steam Generator Structural Integrity (Section 19.2.1.21) (Existing).
22. Structures Monitoring Program (Section 19.2.1.22) (Existing - Requires Enhancement).
23. Tank Inspection Program (Section 19.2.1.23) (Existing - Requires Enhancement).
24. Work Control Process (Section 19.2.1.24) (Existing - Requires Enhancement).
25. Bolting Integrity Program (Section 19.2.1.25) (Existing)

19.1.2 TIME LIMITED AGING ANALYSES AGING MANAGEMENT PROGRAMS:

1. Electrical Equipment Qualification (Section 19.4.1) (Existing).
2. Metal Fatigue of Reactor Coolant Pressure Boundary (Section 19.4.2) (Existing).

19.1.3 REFERENCES FOR SECTION 19.1

- 19.1-1 NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U. S. Nuclear Regulatory Commission, July 2001.

19.2 PROGRAMS THAT MANAGE THE EFFECTS OF AGING ON STRUCTURES AND COMPONENTS WITHIN THE SCOPE OF LICENSE RENEWAL

This section provides summaries of the programs credited for managing the effects of aging on structures and components within the scope of license renewal.

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in NUREG-1800, Section A.2. The Quality Assurance program includes the elements of corrective action, confirmation process, and administrative controls and is applicable to the safety-related and non-safety-related structures, and components that are within the scope of license renewal.

19.2.1 AGING MANAGEMENT PROGRAMS

19.2.1.1 Battery Rack Inspections

Program Description

Battery Rack Inspections is a plant-specific program that manages the aging effect of loss of material. The structural integrity of the support racks for the station batteries, within the scope of license renewal, is verified by visually inspecting for loss of material.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inclusion of In-Scope Battery Racks

The existing inspection program will be modified to include those battery racks that require monitoring for license renewal, but are not already included in the program. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 1.

- Inspection Criteria

Implementing procedures will be modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 2.

19.2.1.2 Boric Acid Corrosion

Program Description

Boric Acid Corrosion corresponds to NUREG-1801, Section XI.M10 “Boric Acid Corrosion.” The program manages the aging effect of loss of material and ensures that systems, structures, and components susceptible to boric acid corrosion are properly monitored. The program uses visual inspections to detect the boric acid leakage source, path, and any targets of the leakage. It ensures that boric acid corrosion is consistently identified, documented, evaluated, trended, and effectively repaired. The Boric Acid Corrosion program provides both detection and analysis of leakage of borated water inside containment. The General Condition Monitoring program is the primary method for detecting borated water leakage outside containment. The analysis of the leakage is performed through the Boric Acid Corrosion program. Any necessary corrective actions are implemented through the Corrective Action Program.

Boric Acid Corrosion program implements the requirements of:

- NRC Bulletin 2001-01 (Reference 19.2-15).
- NRC Bulletin 2002-01 (Reference 19.2-16).
- NRC Bulletin 2002-02 (Reference 19.2-17).
- NRC Bulletin 2003-02 (Reference 19.2-18).
- NRC Order EA-03-009 (Reference 19.2-19).
- NRC Bulletin 2004-01 (Reference 19.2-20).

The acceptance criterion is the absence of any boric acid leakage or precipitation. If boric acid leakage or precipitation is found by any personnel, it is required to be reported using the Corrective Action Program. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.3 Buried Pipe Inspection Program

Program Description

The Buried Pipe Inspection Program is an existing program that corresponds to NUREG-1801, Sections XI.M28, “Buried Piping and Tanks Surveillance” and XI.M34, “Buried Piping and Tanks Inspection.” The program manages the aging effect of loss of material through the use of preventive measures and inspections. The inspections will be performed when the piping and components are excavated for maintenance or for any other reason.

There are no buried tanks within the scope of license renewal.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. In addition to visual inspections, the field inspections for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- **Baseline Inspection**

A baseline inspection of the in-scope buried piping located in a damp soil environment will be performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching will be performed by visual, and mechanical or other appropriate methods. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 3,.

- **Buried Piping Inspections**

The maintenance and work control procedures will be revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures will include the inspection for the loss of material due to selective leaching, which will be performed by visual, and mechanical or other appropriate methods. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 4.

19.2.1.4 Chemistry Control for Primary Systems Program

Program Description

Chemistry Control for Primary Systems Program corresponds to NUREG-1801, Section XI.M2, "Water Chemistry." The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, fluorides, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-105714 (Reference 19.2-1) for primary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality

Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.5 Chemistry Control for Secondary Systems Program

Program Description

Chemistry Control for Secondary Systems Program corresponds to NUREG-1801, Section XI.M2, “Water Chemistry.” The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, sodium, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-102134 (Reference 19.2-2) for secondary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.6 Closed-Cycle Cooling Water System

Program Description

Closed-Cycle Cooling Water System corresponds to NUREG-1801, Section XI. M21, “Closed-Cycle Cooling Water System.” The program manages the aging effect of loss of material through the maintenance of process fluid chemistry and performance monitoring of closed-cycle cooling water systems to ensure parameters remain within acceptable limits. The program is based directly on guidance contained in EPRI Report TR-107396 (Reference 19.2-3).

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following commitment will be implemented prior to the period of extended operation:

- Heat Exchanger Baseline Inspection

A baseline visual inspection will be performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one:

- Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger,

- Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and
- Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 30.

19.2.1.7 Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” as modified by NRC Interim Staff Guidance-05 (Reference 19.2-4). This program manages the aging effects of cracking and embrittlement to ensure that electrical cables, connectors, and fuse holders within the scope of license renewal that are exposed to an adverse localized environment (but not subject to the environmental qualification requirements of 10 CFR 50.49) are capable of performing their intended function. Adverse localized environments may be caused by heat, radiation or moisture.

The acceptance criterion for the visual inspections of accessible non-EQ cable jackets and connector coverings is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following actions will be implemented prior to the period of extended operation:

- Program Implementation

The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be established.

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 5.

- Inclusion of In-Scope Fuse Holders

Fuse holders meeting the requirements will be evaluated prior to the period of extended operation for possible aging effects requiring management. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The Electrical Cables and Connectors Not Subject to 10 CFR 50.49

Environmental Qualification Requirements (if needed for fuse holders) will consider the aging stressors for the metallic clips.

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 6.

19.2.1.8 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Program Description

Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits corresponds to NUREG-1801, Section XI.E2, “Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” and the program as modified in draft NRC ISG-15 (Reference 19.2-5). This program manages the aging effects of cracking and embrittlement for electrical cables within the scope of license renewal that are used in circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation (but not subject to the environmental qualification requirements of 10 CFR 50.49), and are installed in adverse localized environments caused by heat, radiation or moisture.

The acceptance criterion for the calibration readings is the loop-specific tolerances established in Technical Specifications and surveillance procedures. Where calibration of the instrumentation is not performed in situ, the acceptance criteria for each test are defined by the specific type of test performed and the specific cable tested. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Testing of Cables for Instruments That Are Not Calibrated In Situ

Procedures will be developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ. The first tests will be completed prior to the period of extended operation. The frequency of subsequent tests will be based on Engineering evaluation and will not exceed a 10 year interval. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 7.

- Review of Surveillance Test Results for Cables Tested In Situ

Calibration results for cables tested in situ will be reviewed to detect severe aging degradation of the cable insulation. The initial review will be completed prior to the period

of extended operation and will include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 33.

19.2.1.9 Fire Protection Program

Program Description

The Fire Protection Program is an existing program and corresponds to NUREG-1801, Sections XI.M26, “Fire Protection” and XI.M27, “Fire Water System” and to the revised XI.M27, “Fire Water System” program described in NRC Interim Staff Guidance (ISG)-04 (Reference 19.2-6). The program manages the aging effects of loss of material, cracking, and change of material properties for plant fire protection features and components. The program manages these aging effects through the use of periodic inspections and tests.

The program also manages the aging effects for the diesel-driven fire pump fuel supply line, the reactor coolant pump oil collection systems, and Appendix R support equipment.

Visual inspection of fire protection piping internal surfaces that are exposed to water is performed when the system is opened for maintenance and/or repair. The Work Control Process provides guidance for the performance of internal inspections of fire protection piping and components whenever the system is opened for maintenance or repair.

The acceptance criteria for the Fire Protection Program are:

- For visual inspections, the absence of anomalous indications that are signs of degradation.
- For fire barriers and fire doors, the sizes for breaks, holes, cracks, spalling gaps, and/or clearances are in accordance with the limits established in the inspection procedures.
- For fire protection equipment performance tests (i.e., flow and pressure tests), acceptance criteria are provided in the appropriate surveillance procedures.

Additionally, the fire protection water system pressure is continuously monitored to be above the minimum setpoint. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Baseline Fire Protection Inspections

A baseline visual inspection will be performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 8.

The following program enhancement will be implemented prior to the sprinkler heads achieving 50 years of service life:

- Testing or Replacement of Sprinkler Heads

Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years will be included in the Fire Protection Program. The first tests will be completed prior to the sprinkler heads achieving 50 years of service life. The frequency of subsequent tests will not exceed a 10 year interval. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 9.

19.2.1.10 Flow-Accelerated Corrosion

Program Description

Flow-Accelerated Corrosion Program corresponds to NUREG-1801, Section XI.M17, “Flow-Accelerated Corrosion.” The program manages the aging effect of loss of material in accordance with the EPRI guidelines in NSAC-202L (Reference 19.2-7). It includes procedures or administrative controls to assure that the structural integrity of carbon steel and low-alloy steel piping and components, such as valves, steam traps, and feedwater heaters, is maintained.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.11 Fuel Oil Chemistry

Program Description

Fuel Oil Chemistry corresponds to NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.” The program manages the aging effect of loss of material by monitoring and controlling fuel oil quality to ensure that it is compatible with the materials of construction for in-scope components containing diesel fuel oil.

The Fuel Oil Chemistry program uses the following industry standards as the basis for the program:

- ASTM Standard D 1796 (Reference 19.2-8),

- ASTM Standard D 2276 (unmodified) (Reference 19.2-9), and
- ASTM Standard D 4057 (Reference 19.2-10).

The acceptance criterion is adherence to the specific guidelines and limits defined in related plant procedures for parameters that have been shown to contribute to component degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.12 General Condition Monitoring

Program Description

General Condition Monitoring is a plant-specific program that manages the aging effects of loss of material, change of material properties, and cracking on the external surfaces of components. It is performed in accessible plant areas for components and structures including those within the scope of license renewal and involves visual inspections for evidence of age-related degradation. General Condition Monitoring is implemented by Radiation Protection technicians, System Engineers, and Plant Equipment Operators while performing their routine in-plant activities.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Procedure and Training Enhancements

The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns will be enhanced to provide expectations that identify the requirements for the inspection of aging effects. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 10.

19.2.1.13 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E3, “Inaccessible Medium-Voltage Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements.” This program manages the aging effect of formation of water trees and ensures that inaccessible medium-voltage (2 kV to 15 kV) electrical cables within the scope of license renewal (but not subject to the environmental qualification requirements of 10 CFR 50.49) that have been submerged, remain capable of performing their intended function. The program considers the combined effects of submergence, simultaneous with a significant voltage exposure. Significant voltage exposure is defined as being subjected to system voltage for more the twenty-five percent of the time.

The acceptance criterion for the inspections performed under the Structures Monitoring Program is to confirm that in-scope, medium-voltage cables have not become submerged. In-scope cable found to be submerged in standing water for an extended period of time will be subject to an engineering evaluation and corrective action. The evaluation will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The test will use a proven methodology for detecting deterioration of the insulation due to wetting. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Verification Testing

In-scope cable found to be submerged will be subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation will also address the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test will use a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables, or other appropriate testing. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into

the Corrective Action Program. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 11.

- Testing of Inaccessible Medium Voltage Cables

The in-scope cables in Unit 3 duct lines # 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and # 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C and 35D) have been tested to demonstrate that water treeing will not prevent the cables from performing their intended function. Subsequent testing has been scheduled to be performed on a frequency not to exceed a 10 year interval.

This completes the actions required to complete commitment Item 34. in Table 19.6 - 1, License Renewal Commitments.

- Sample Testing of Inaccessible Medium Voltage Cables

Prior to the period of extended operation, a representative sample of in-scope medium voltage cables will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function. This sample testing is in addition to the testing specified in the previous commitment. Subsequent testing will be performed on a frequency not to exceed a 10-year interval. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 35.

19.2.1.14 Infrequently Accessed Areas Inspection Program

Program Description

Infrequently Accessed Areas Inspection Program is a plant-specific program that manages the aging effects of loss of material, change of material properties, and cracking. The program uses visual inspections of the external surfaces of in-scope structures and components located in infrequently accessed areas of the plant.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Program Implementation

The Infrequently Accessed Areas Inspection Program will be established.

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 12.

19.2.1.15 Inservice Inspection Program: Containment Inspections

Program Description

Inservice Inspection Program: Containment Inspections corresponds to the following NUREG-1801 program descriptions:

- Section XI.S1, “ASME Section XI, Subsection IWE”,
- Section XI.S2, “ASME Section XI, Subsection IWL”, and
- Section XI.S4, “10 CFR Part 50, Appendix J.”

The program manages the aging effects of loss of material, change of material properties, and cracking. The program is consistent with ASME Section XI, Subsections IWE and IWL, and 10 CFR 50.55a(b)(2), which provide the criteria for ISI Containment inspections.

Appendix J Leakage Rate Testing is included as part of the Inservice Inspection Program: Containment Inspections. The Containment Appendix J Leakage Rate Test Program implements Type A and B tests to measure the overall primary Containment integrated leakage rate.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Containment Inspections are based on the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.16 Inservice Inspection Program: Reactor Vessel Internals

Program Description

Inservice Inspection Program: Reactor Vessel Internals corresponds to the following NUREG-1801 program descriptions:

- Section XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”.
- Section XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”.
- Section XI.M16, “PWR Vessel Internals”.

The Inservice Inspection Program: Reactor Vessel Internals manages the effects of aging for those reactor internals that are susceptible to loss of material, cracking, loss of pre-load, change in dimension and loss of fracture toughness (which presents itself as cracking due to embrittlement).

Industry groups are in place whose objectives include the investigation of the aging effects applicable to reactor vessel internals regarding such items as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC and IASCC), and loss of pre-load for baffle and former-assembly bolts.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Reactor Vessel Internals are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be implemented at least two years prior to period of extended operation:

- Reactor Vessel Internals Inspections

Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC and IASCC), and loss of pre-load for baffle and former-assembly bolts and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, will be submitted to the NRC for approval.

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 13.

The following program enhancement will be implemented at least two years prior to the period of extended operation:

- Augmented Holddown Spring Inspections

Augmented inspection of the Millstone Unit 3 core barrel holddown spring will be performed. In particular, the inspection will detect gross indication of loss of preload as an aging effect. As an alternative to performing an augmented inspection, the holddown spring will be replaced. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 14.

19.2.1.17 Inservice Inspection Program: Systems, Components and Supports

Program Description

Inservice Inspection Program: Systems, Components and Supports corresponds to the following NUREG-1801 program descriptions:

- Section XI.M1, “ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD”,
- Section XI.M3, “Reactor Head Closure Studs”,
- Section XI.M11, “Ni-Alloy Nozzles and Penetrations”,
- Section XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”, and
- Section XI.S3, “ASME Section XI, Subsection IWF.”

The Inservice Inspection Program: Systems, Components and Supports is an existing program that was developed to comply with the requirements of ASME Boiler and Pressure Vessel Code, Section XI (Reference 19.2-11). The ASME program provides the requirements for ISI, repair, and replacement for all Class 1, 2 and 3 components and the associated component supports. For license renewal, the Millstone program has been credited to manage the effects of aging for only Class 1 and specific Class 2 components (on the secondary side of the steam generators as determined through the aging management review process) and for Class 1, 2, and 3 components supports. Inservice Inspection Program: Systems, Components and Supports manages the aging effects of cracking, loss of fracture toughness, loss of material and loss of preload.

Industry programs are in place whose objectives include the investigation of aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identification of appropriate aging management activities.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Systems, Components and Supports are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be taken prior to the period of extended operation:

- PWSCC of Nickel-Based Alloys

Millstone will follow the industry efforts investigating the aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description will be submitted at least two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 50.54.21(a)(3). This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 15.

- **Monitoring Fracture Toughness**

For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally-embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 28,.

- **Pressurizer Spray Head Assembly Cracking**

The pressurizer spray head assembly will be either replaced or inspected utilizing the best currently available (at the time of inspection) techniques for detecting cracking resulting from SCC. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 37.

19.2.1.18 Inspection Activities: Load Handling Cranes and Devices

Program Description

Inspection Activities: Load Handling Cranes and Devices corresponds to NUREG-1801, Section XI. M23, “Inspection of Overhead Heavy Load (Related to Refueling) Handling Systems.” The program manages the aging effect of loss of material for the load handling cranes and devices within the scope of license renewal. The in-scope load handling cranes and devices are either safety-related or seismically designed to ensure that they will not adversely impact safety-related components during or subsequent to a seismic event.

Inspection Activities: Load Handling Cranes and Devices addresses the overall condition of the crane or device, including checking the condition of the structural members (i.e., rails, girders, etc.) and fasteners on the crane or device, the runways along which the crane or device moves, and the baseplates and anchorages for the runways and monorails.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inclusion of In-Scope Lifting Devices

The existing inspection program will be modified to include those lifting devices that require monitoring for license renewal, but are not already included in the program. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 16.

- Inspection Criteria

Implementing procedures and documentation will be modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 17.

19.2.1.19 Reactor Vessel Surveillance

Program Description

Reactor Vessel Surveillance corresponds to NUREG-1801, Section XI.M31 “Reactor Vessel Surveillance”. The Reactor Vessel Surveillance program manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low-alloy subcomponents in the beltline region of the reactor vessel. Neutron dosimetry and material properties data derived from the reactor vessel materials’ irradiation surveillance program are used in calculations and evaluations that demonstrate compliance with applicable regulations. This program ensures compliance with Technical Requirements Manual requirements that surveillance specimens are removed and examined at predetermined intervals established in the Technical Specification to monitor the changes in the material properties and the results of the examinations used to update the Technical Specification operating limits.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.20 Service Water System (Open-Cycle Cooling)

Program Description

The Service Water System (Open-Cycle Cooling) program corresponds to NUREG-1801, Section XI.M20, “Open Cycle Cooling Water System.” The program manages the aging effects of loss of material and buildup of deposits. The program implements the NRC guidelines in Generic

Letter 89-13 (Reference 19.2-12), which includes (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion (including microbiologically influenced corrosion), erosion, protective coating failure, silting, and biofouling do not degrade the performance of safety-related systems serviced by Service Water System; (d) a system walkdown inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures. Millstone Unit 3 relies on either frequent, regular inspection and cleaning of heat exchangers, or maintaining of heat exchangers in dry or fresh water lay-up to preclude fouling.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.1.21 Steam Generator Structural Integrity

Program Description

Steam Generator Structural Integrity corresponds to NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity Program.” This program manages the aging effects of loss of material and cracking and adopts the performance criteria and guidance for monitoring and maintaining steam generator tubes as defined in NEI 97-06 (Reference 19.2-13). The program incorporates performance criteria for structural integrity, accident-induced leakage, and operational leakage. The program includes preventive measures to mitigate degradation through the control of primary and secondary side water chemistry; assessment of degradation mechanisms; inservice inspection of the steam generator tubes to detect degradation; evaluation and plugging or repair, as needed; and leakage monitoring to ensure the structural and leakage integrity of the pressure boundary.

Industry programs are in place whose objectives include the investigation of aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identification of appropriate aging management activities.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be implemented based on the availability of the industry guidance:

- PWSCC of Nickel-Based Alloys

Millstone will follow the industry efforts investigating the aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description will be submitted prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 50.54.21(a)(3).

This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 15.

19.2.1.22 Structures Monitoring Program

Program Description

Structures Monitoring Program corresponds to the following NUREG-1801 program descriptions:

- Section XI.S5 “Masonry Wall Program”,
- Section XI.S6 “Structures Monitoring Program”, and
- Section XI.S7 “R.G. 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants”.

The Structures Monitoring Program manages the aging effects of loss of material, change of material properties, and cracking by the monitoring of structures and structural support systems that are in the scope of license renewal. The majority of these structures and structural support systems are monitored under 10 CFR 50.65 (Reference 19.2-14). Other structures in the scope of license renewal (such as non-safety related buildings and enclosures, duct banks, valve pits and trenches, HELB barriers, and flood gates) are also monitored to ensure there is no loss of intended function.

The scope includes all masonry walls and water-control structures identified as performing intended functions in accordance with 10 CFR 54.4.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Modification of Structures Monitoring Program procedures

NUREG-1801 recommends the use of ACI 349.3R-96 and ANSI/ASCE 11-90, as a reference for recommendations for the development of an evaluation procedure for nuclear safety-related concrete structures and existing buildings. These documents were not used or referenced as a standard for establishing the Structures Monitoring Program. The implementing procedures will be modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 18.

- Addition of Structures to the Structures Monitoring Program

The Structures Monitoring Program does not currently monitor all structures in-scope for license renewal. The Structures Monitoring Program and implementing procedures will be modified to include all in-scope structures. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 19.

- Sampling of Groundwater

Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 20.

- Engineering Notification of Submerged Medium Voltage Cables

The Structures Monitoring Program and implementing procedures will be modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 21.

- Inspection of Normally Inaccessible Areas That Become Accessible

The maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 22.

19.2.1.23 Tank Inspection Program

Program Description

Tank Inspection Program corresponds to NUREG-1801, Section XI.M29, "Aboveground Carbon Steel Tanks." The program manages the aging effect of loss of material through periodic internal and external tank inspections. The program includes inspections of the sealant and caulking in and around the tank and the concrete foundation and evaluations to monitor the condition of coatings,

linings, and structural elements, to prevent deterioration of the tanks to unacceptable levels. The program also includes volumetric examination of inaccessible locations, such as the external surfaces of tank bottoms.

The acceptance criterion for visual inspections of paint, coatings, sealant, caulking, and structural elements is the absence of anomalous indications that are signs of degradation. Thickness measurements of the tank walls and bottoms are evaluated against design thickness, established baseline values, or loss of material allowances. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inspection of sealants and caulking

Appropriate inspections of sealants and caulking used for moisture intrusion prevention in and around aboveground tanks will be performed. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 23.

- Non-destructive Volumetric Examination of Inaccessible Tank Bottoms

Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, will be performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 24.

- Tanks Being Added to Tank Inspection Program

The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and have been included in the respective Tank Inspection Program inspection plan. These changes complete the action required for commitment Item 25 in Table 19.6 - 1.

19.2.1.24 Work Control Process

Program Description

Work Control Process is a plant specific program that integrates and coordinates the combined efforts of Maintenance, Engineering, Operations, and other support organizations to manage maintenance activities. The Work Control Process is utilized to manage the aging effects of loss of material, change of material properties, cracking, and buildup of deposits for components and

plant commodities within the scope of license renewal. Performance testing and maintenance activities, both preventive and corrective, are planned and conducted in accordance with the Work Control Process. The Work Control Process also provides opportunities to collect oil and engine coolant fluid samples for subsequent analysis of contaminants and chemical properties, which could either indicate or affect aging.

In addition to visual inspections, the field inspection for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping.

The acceptance criterion for visual inspections is the absence of anomalous signs of degradation. The acceptance criteria for testing or sampling are specified in the various station procedures and/or vendor technical manuals or recommendations. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- **Performance of Inspections During Maintenance Activities**

Changes will be made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 26.

- **Selective Leaching Inspection**

Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching will be performed on a representative sample of locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 31.

- **Verification of Program Scope**

A review of the Work Control Process inspection opportunities for each material and environment group supplemental to the initial review, supplemental to the initial review conducted during the development of the LRA, will be performed. Baseline inspections will be performed for the material and environment combinations that have not been inspected as part of the Work control Process. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 32.

19.2.1.25 Bolting Integrity Program

Program Description

The Bolting Integrity Program corresponds to NUREG-1801, Section XI.M18, “Bolting Integrity”. The program manages the aging effects of cracking, loss of material and loss of preload.

This is accomplished by establishing good bolting practices in accordance with EPRI NP-5067, Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual, and Volume 2: Small Bolts and Threaded Fasteners and EPRI TR-104213, Bolted Joint Maintenance and Application Guide. For ASME Class bolting, aging effects are additionally managed by the performance of inservice examinations in accordance with ASME Section XI, Subsections IWB, IWC, IWD, and IWF.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.2.2 REFERENCES FOR SECTION 19.2

- 19.2-1 TR-105714, PWR Primary Water Chemistry Guidelines, Technical Report, Revision 3, Electric Power Research Institute.
- 19.2-2 TR-102134, PWR Secondary Water Chemistry Guidelines, Technical Report, Revision 3, Electrical Power Research Institute.
- 19.2-3 EPRI TR-107396, Closed Cooling Water Chemistry Guideline, Technical Report, Electrical Power Research Institute, Palo Alto, CA, November 1997.
- 19.2-4 NRC Interim Staff Guidance (ISG)-05, The Identification And Treatment of Electrical Fuse Holders For License Renewal, U.S. Nuclear Regulatory Commission, March 10, 2003.
- 19.2-5 Letter from Pao-Tsin Kuo, Nuclear Regulatory Commission, to Alex Marion, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, Proposed Interim Staff Guidance (ISG)-15: Revision of Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) X1.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits”, August 12, 2003.
- 19.2-6 NRC Interim Staff Guidance (ISG)-04, “Aging Management of Fire Protection Systems for License Renewal”, U.S. Nuclear Regulatory Commission, December 3, 2002.

- 19.2-7 NSAC-202L-R4, Recommendations for an Effective Flow Accelerated Corrosion Program, Electric Power Research Institute, November, 2013.
- 19.2-8 ASTM D 1796, Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method, American Society for Testing Materials, West Conshohocken, PA.
- 19.2-9 ASTM D 2276, Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling, American Society for Testing Materials, West Conshohocken, PA.
- 19.2-10 ASTM D 4057, Standard Practice for Manual Sampling of Petroleum and Petroleum Products, American Society for Testing Materials, West Conshohocken, PA.
- 19.2-11 ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers.
- 19.2-12 Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment, Nuclear Regulatory Commission, July 18, 1989 (Supplement 1 dated 4/4/90).
- 19.2-13 NEI 97-06, Steam Generator Program Guidelines, Technical Report, Nuclear Energy Institute.
- 19.2-14 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, U. S. Nuclear Regulatory Commission.
- 19.2-15 NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, U.S. Nuclear Regulatory Commission, August 3, 2001.
- 19.2-16 NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, March 18, 2002.
- 19.2-17 NRC Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs, U.S. Nuclear Regulatory Commission, August 9, 2002.
- 19.2-18 NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, 08/21/03.
- 19.2-19 NRC Order EA-03-009, Issuance of Order Establishing Interim Inspection Requirements For Reactor Pressure Vessel Heads At Pressurized Water Reactors, U.S. Nuclear Regulatory Commission, February 11, 2003.”
- 19.2-20 NRC Bulletin 2004-01, Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, May 28, 2004.

19.3 TIME-LIMITED AGING ANALYSIS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

19.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in FSAR Section 5.3. Time-limited aging analyses (TLAAs) applicable to the reactor vessel are:

- Upper-shelf energy.
- Pressurized thermal shock.
- Pressure-temperature limits.

The Reactor Vessel Surveillance program manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time-dependent parameters used in the aging analyses for pressurized thermal shock, upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

The reactor vessel neutron embrittlement evaluations have been based on 54 effective full power years of operation. 54 effective full power years would be reached at the end of the period of extended operation (60 years) assuming a capacity factor of 90% for the lifetime of the unit.

19.3.1.1 Upper Shelf Energy

10 CFR 50, Appendix G contains screening criteria that establish limits on how far the upper shelf energy values for a reactor pressure vessel material may be allowed to drop due to neutron irradiation exposure. The regulation requires the initial upper shelf energy value to be greater than 75 ft-lbs in the unirradiated condition and for the value to be greater than 50 ft-lbs in the fully irradiated condition as determined by Charpy V-notch specimen testing throughout the licensed life of the plant. Upper shelf energy values of less than 50 ft-lbs may be acceptable to the NRC if it can be demonstrated that these lower values will provide margins of safety against brittle fracture equivalent to those required by ASME Section XI, Appendix G.

Acceptable upper shelf energy values have been calculated in accordance with Regulatory Guide 1.99, Revision 2 to the end of the period of extended operation. Calculated upper shelf energy values for the most limiting reactor pressure vessel beltline plate and weld materials remain greater than 50 ft-lbs.

19.3.1.2 Pressurized Thermal Shock

Reactor pressure vessel beltline fluence is one of the factors used to determine the margin to reactor pressure vessel pressurized thermal shock as a result of radiation embrittlement. The

margin is the difference between the maximum nil ductility reference temperature in the limiting beltline material (RT_{PTS}) and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forging and axial weld materials, and 300°F for beltline circumferential weld materials.

Acceptable RT_{PTS} values have been calculated in accordance with Regulatory Guide 1.99, Revision 2, requirements to the end of the period of extended operation.

19.3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits identify the maximum allowable pressure as a function of reactor coolant temperature. As the pressure vessel becomes irradiated and its fracture toughness is reduced, the allowable pressure at low temperatures is reduced. Therefore, in order to heatup and cooldown the Reactor Coolant System, the reactor coolant temperature and pressure must be maintained within the limits of Appendix G as defined by the reactor vessel fluence.

Heatup and cooldown limit curves have been calculated using the adjusted RT_{NDT} corresponding to the limiting beltline material of the reactor pressure vessel for the current period of licensed operation. Current cold overpressure protection system (COPS) heatup and cooldown limit curves were approved in License Amendment 197.

In accordance with 10 CFR 50, Appendix G, updated pressure-temperature limits for entering the period of extended operation will be developed and implemented prior to the period of extended operation. Cold overpressure protection system enable temperature requirements will be updated to ensure that the pressure-temperature limits will not be exceeded for postulated plant transients during the period of extended operation. Millstone Unit 3 will calculate USE, RT_{PTS} , and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation.

19.3.2 METAL FATIGUE

Fatigue is defined as structural deterioration that can occur through repeated stress or strain cycles resulting from fluctuations in loads and/or temperatures. After repeated cyclic loading of sufficient magnitude, micro-structural damage can accumulate leading to microscopic crack initiation at the most highly affected locations. Fatigue cracks typically initiate at points of maximum local stress ranges and minimum local strength. Further cyclic mechanical and/or thermal loading can lead to crack growth.

Fatigue represents an aging mechanism. As such, fatigue evaluations represent a time-limited aging analysis even though the system, structure and component design limits are based upon the

number of cycles and the associated fatigue (cumulative) usage factors rather than specific time limits.

19.3.2.1 Millstone Unit 3 Class 1 Components

Components within the Millstone Unit 3 nuclear steam supply system are subject to a wide variety of varying mechanical and thermal loads that contribute to fatigue accumulation. The Reactor Coolant System components are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III (Reference 19.3-21) this code requires that design analyses for Class 1 systems and components address fatigue and the establishment of load limits to preclude initiation of fatigue cracks.

The type and number of Reactor Coolant System design transients have been identified. In all instances, the number of Reactor Coolant System design transients assumed in the original design were found to be acceptable for the period of extended operation.

NRC Bulletin 88-08 identified a concern regarding potential temperature stratification or temperature oscillations in unisolable sections of piping attached to the Reactor Coolant System. Based upon the Millstone Unit 3 response (Reference 19.3-22) and supplemental communications, the NRC concluded that Millstone Unit 3 meets the requirements of Bulletin 88-08 (Reference 19.3-23).

Pressurizer surge line thermal stratification was a concern raised by the NRC in Bulletin 88-11. One of the requirements of this bulletin was to analyze the effects of thermal stratification on surge line integrity. These analyses were collectively performed as a Westinghouse Owners Group task (Reference 19.3-24) supplemented by additional unit specific inspections and activities. Based upon the Westinghouse Owners Group task, the NRC concluded that the bounding evaluations and supplemental unit specific inspections and activities demonstrate that the Millstone Unit 3 pressurizer surge line piping and associated nozzles meet Bulletin 88-11 requirements (References 19.3-25, and 19.3-26, and 19.3-27). The NRC has reviewed this information and determined that Millstone Unit 3 has addressed the actions required by Bulletin 88-11 (Reference 19.3-28).

Thermal aging refers to changes in the microstructure and properties of a susceptible material due to prolonged exposure to elevated temperatures above approximately 480°F. Reactor Coolant System temperatures exceed this threshold. At these temperatures, the hardness of potentially susceptible Cast Austenitic Stainless Steel (CASS) materials increase while their ductility, impact strength and more importantly, their fracture toughness, decrease. Fracture toughness is one of the more important design inputs in a leak-before-break and a flaw tolerance evaluation, performed to ensure protection of the reactor coolant system against guillotine pipe breaks throughout plant life. The degree of change in fracture toughness (thermal embrittlement) is dependent on the time of exposure to these elevated temperatures.

Except for the pressurizer spray head, acceptable thermal and pressure transients, and operating cycles have been projected for ASME Section III, Class 1 components, through the period of

extended operation. Thermal aging of the pressurizer spray head will be managed through the period of extended operation.

Commitments

The following actions will be implemented prior to the period of extended operation:

Thermal aging of the pressurizer spray head will be managed by the Inservice Inspection Program: Systems, Components and Supports. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 28.

19.3.2.2 Non-Class 1 Components

Non-Class 1 components can include ASME Section III Classes 2 and 3, ANSI Standard B31.7 Classes 2 and 3, and ANSI Standard B31.1 (Reference 19.3-29) piping and tubing. Piping systems designed to these requirements (e.g., sample lines) incorporate a stress range reduction factor to conservatively address the effects of thermal cycling on fatigue. For those sample lines projected to experience greater than 7,000 equivalent full-temperature thermal cycles, actual expansion stresses did not exceed allowable expansion stresses.

Acceptable numbers of thermal cycles and acceptable expansion stresses have been projected to the end of the period of extended operation.

19.3.2.3 Environmentally Assisted Fatigue

The effect of reactor coolant environment on fatigue is generally referred to as environmentally assisted fatigue. As part of an industry effort to address environmental effects on operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories evaluated fatigue-sensitive component locations at plants designed by all four domestic nuclear steam supply system vendors. These evaluations are presented and discussed in NUREG/CR-6260 (Reference 19.3-30). The evaluations associated with the newer-vintage Westinghouse plants are applicable, since the majority of the Millstone Unit 3 Class 1 systems and components were designed to ASME Section III requirements.

The influence of the reactor water environment on the cumulative usage factor was evaluated for the following representative components identified in NUREG/CR-6260 for the period of extended operation, using the most recent laboratory data and methods:

- Reactor vessel shell and lower head.
- Reactor vessel inlet and outlet nozzles.
- Surge line.
- Charging nozzle.

- Safety Injection System nozzle.
- Residual Heat Removal System Class 1 piping.

These six fatigue-sensitive locations have been evaluated using the methods identified in NUREG/CR-6583 (Reference 19.3-31), and NUREG/CR-5704 (Reference 19.3-32).

Utilizing Millstone Unit 3 cyclic and transient information, four fatigue sensitive component locations were determined to have cumulative usage factors (CUFs) greater than 1.0 over the period of extended operation. For the pressurizer surge line, charging nozzle, safety injection nozzles, and Residual Heat Removal System piping, more detailed stress analyses or fatigue monitoring and cycle counting would have to be used to reduce CUF below 1.0. Due to code conservatisms included in the ASME Code, a CUF of greater than 1.0 does not mean that fatigue cracking will occur; only that there is a potential for fatigue cracking to occur over the period of extended operation. Utilizing these conservatisms, an approach will be developed to manage the effects of environmentally assisted fatigue for those specific locations with a CUF greater than 1.0. The expected approach is to manage these effects through the use of an inspection program that has been reviewed and approved by the NRC. The program would be expected to include, for example, appropriate non-destructive examinations and NRC acceptable inspection periods. Repair or replacement activities would be based upon inspection results.

Commitments

The following actions will be implemented prior to the period of extended operation:

- The effects of environmentally assisted fatigue for those specific locations with a CUF greater than 1.0 will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary program for the period of extended operation. If the specific locations are not repaired, replaced, or successfully re-analyzed, a modified inspection program description, including a comparison to the 10 program elements of NUREG-1801 program, will be submitted to the NRC for approval. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 27.
- Millstone will follow industry efforts that will provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will also implement the appropriate recommendations resulting from this guidance. Until these recommendations are available, Millstone 3 commits to using the pressurizer surge line nozzle as a leading indicator to address environmental effects on fatigue of pressurizer sub-components during the period of extended operation. This commitment is identified in Table 19.6 - 1, License Renewal Commitments, Item 29.

19.3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Electrical Equipment Qualification (EEQ) program is an integral part of the design, construction and operation of nuclear power generating stations. A description of this program provided in Section 19.3, Environmental Qualification (EQ) of Electric Equipment.

10 CFR Part 50 requires that certain categories of systems, structures and components be designed to accommodate the effects of both normal and accident environmental conditions, and that design control measures be employed to ensure the adequacy of these designs. Specific requirements pertaining to the environmental qualification of these categories of electrical equipment are embodied within 10 CFR 50.49 (Reference 19.3-33). The categories include safety-related (Class 1E) electrical equipment, non-safety-related electrical equipment whose failure could prevent the satisfactory accomplishment of a safety function by safety-related equipment, and certain post-accident monitoring equipment. As required by 10 CFR 50.49, electrical equipment not qualified for the current license term is to be refurbished, replaced or have its qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical equipment that specify a qualification of 40 years or greater are considered to represent a time-limited aging analysis. Guidance relating to the methods and procedures for implementing the requirements of 10 CFR 50.49 is contained within Regulatory Guide 1.89 (Reference 19.3-34). Further guidance for post-accident monitoring equipment is contained within Regulatory Guide 1.97 (Reference 19.3-35).

Environmental qualification of electrical equipment will be adequately managed for the period of extended operation.

19.3.4 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

19.3.4.1 Containment Liner Plate

Millstone Unit 3 has a conventionally reinforced concrete Containment structure maintained at subatmospheric pressure, surrounded by an enclosure building. A welded carbon steel liner plate is attached to the inside surface of the concrete, providing a high degree of leak tightness. Components of the liner plate include penetration sleeves, access openings, and piping penetrations.

Evaluations of the Containment liner plate involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses. Acceptable Containment liner plate integrity has been projected to the end of the period of extended operation.

19.3.4.2 Containment Penetrations

Millstone Unit 3 Containment penetrations are used for personnel and equipment access, process piping, electrical service, or for a mechanical fuel transfer system. Each of these penetrations is anchored to, and transfers loads to the reinforced Containment wall. There were no applicable

codes for the design of concrete Containment liners at the beginning of the construction of the Millstone Unit 3 liner. ASME Section III, Division 1 and 2, and ASME Section VIII were used as guides.

Evaluations of Containment liner plate components involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses.

Acceptable Containment penetration integrity has been projected to the period of extended operation.

19.3.5 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

19.3.5.1 Crane Load Cycle Limit

The containment polar crane, spent fuel crane, monorails, and jib cranes are examples of the types of cranes within the scope of license renewal. These cranes meet the guidance contained in NUREG-0612.

The evaluation of crane loads represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of a time-limited assumption, load cycles. The most frequently used crane is the spent fuel crane. Considering all uses, the spent fuel crane is expected to conservatively experience a total number of load cycles over a 60-year period, that is well below the number of cycles allowed in Crane Manufacturers Association of America, Inc. Specification No. 70.

Acceptable crane load cycles have been projected to the end of the period of extended operation.

19.3.5.2 Reactor Coolant Pump Flywheel

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of primary coolant flow to the core in the event that pump power is lost. During normal operation, the reactor coolant pump flywheels develop sufficient kinetic energy to produce high-energy missiles in the event of failure. Conditions that may result in overspeed of the pump increase both the potential for failure and the kinetic energy of the flywheel.

Westinghouse Report WCAP-14535A, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination (Reference 19.3-36) presents an evaluation of the likelihood of flywheel failure over a 60-year period of operation and the justification for relaxation of RG 1.14, Revision 1, Regulatory Position C.4.b(1), requirements to those identified in Regulatory Position C.4.b(2). Using this evaluation, the NRC issued Amendment No. 169 to the unit Technical Specifications, consistent with RG 1.14, Revision 1, Regulatory Position C.4.b(2), to allow the examination of each reactor coolant pump flywheel at least once every 10-years, coinciding with the ASME Section XI inservice inspection program schedule.

The evaluation of reactor coolant pump flywheels represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as thermal cycles and crack growth rates. This evaluation, which indicates a low likelihood of flywheel fatigue failure over a 60-year period, along with implementation of the Inservice Inspection Program: Systems, Components and Supports, provides reasonable assurance that flywheel cracking will be adequately managed for the period of extended operation.

Reactor coolant pump flywheel fatigue cracking will be adequately managed for the period of extended operation.

19.3.5.3 Leak-Before-Break

The Leak-Before-Break (LBB) analysis was evaluated as a time-limited aging analysis (TLAA) to determine that the analysis remains valid for the period of extended operation. The reactor coolant system loop piping (hot leg, cold leg and crossover piping) has been evaluated for LBB.

The LBB analysis was determined to remain valid for the period of extended operation by evaluating their time-based inputs. Thermal aging of cast austenitic stainless steel (CASS) materials and fatigue crack growth calculations were determined to be time-based inputs as defined in 10 CFR 54.3 and required evaluation for the period of extended operation.

The metal fatigue TLAA evaluations described in FSAR Section 19.3.2.1 conclude that design basis limits are not exceeded for ASME Class 1 components (which envelopes the components evaluated for LBB) through the period of extended operation.

Thermal aging of CASS materials for components that have been evaluated for LBB has been evaluated as a TLAA since long-term exposure of CASS materials to reactor coolant system operating temperatures results in an increase in material hardness while its ductility, impact strength and fracture toughness decrease. Fracture toughness represents one of the more important design inputs in a LBB evaluation. The degree of reduction in CASS fracture toughness is dependent on the time of thermal exposure. However, the change in material properties due to thermal aging reaches a saturation value, after which material property changes resulting from additional thermal exposure are not significant. The evaluation of the thermal aging of CASS material for the LBB evaluations consisted of a review to determine whether the fracture toughness value used in the analysis was conservative relative to the fully aged value for fracture toughness for the CASS components. The review concluded that the analysis values were either equal to or lower than the worst-case saturation (fully aged) values for fracture toughness in all cases. Therefore, since the CASS material property values used in current design basis LBB evaluations represent fully aged (saturation) values, and since these values would not change with further exposure time, the LBB evaluations remain valid for the thermal aging of CASS materials throughout the period of extended operation.

The LBB analysis has been projected to remain valid through the end of the period of extended operation.

19.3.5.4 Containment Subfoundation

The Unit 3 Containment basemat is 10 feet thick and is supported by a subfoundation, which is founded on bedrock. The subfoundation consists of (from bottom to top): (1) a 10 inch layer of porous concrete made of Portland cement and coarse aggregate, (2) approximately 1/16-inch rubber waterproofing membrane, (3) a 2 inch layer of Portland cement (PC) mortar seal, (4) a 9 inch layer of porous concrete made of calcium aluminate cement and coarse aggregate [High Alumina Cement, or HAC layer], and (5) thin mortar seal.

In 1987, Unit 3 identified cement constituents (calcium-alumina, which forms a white residue) in the drainage system installed in the HAC layer of the Containment subfoundation. An evaluation determined that the rubber waterproofing membrane had developed leaks, which allowed for the ingress of water into the HAC layer.

Core tests and plate bearing tests were conducted, along with additional testing on HAC mock-ups that were built to the same specifications as used in the original construction of the MP3 subfoundation.

Several core samples were removed from the HAC porous concrete layer in the subfoundation of the ESF Building, where a portion of the building subfoundation is the same as that for the containment basemat. Tests were conducted on these samples to quantify the available margin in the bearing stresses below the containment basemat, for the current license period of 40 years.

In 2005, a condition assessment was performed to determine the acceptability of the Unit 3 containment subfoundation porous concrete layers for the period of extended operation. Computation of bearing stresses on the porous cement surface showed that for a bounding loss of even as much as 7.4% in foundation area (or volume), the bearing stress remains significantly less than the tested strength of 2850 psi. The amount of loss in this scenario bounds the projection of the total amount of white residue that is conservatively calculated to be collected from the construction of the plant through the period of extended operation.

The evaluation of the Millstone Unit 3 containment subfoundation represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as the maximum amount of calcium-alumina that can be leached over time from the HAC layer and still maintain adequate support for the containment basemat.

The structural integrity of the Millstone 3 (MP3) Containment subfoundation has been demonstrated through the period of extended operation. Consistent with 10 CFR 54.21(c)(1), Option (ii), the analyses have been projected to the end of the period of extended operation.

19.3.6 REFERENCES FOR SECTION 19.3

19.3-21 ASME Section III, "Rules for Construction of Nuclear Vessels", ASME Boiler and Vessel Pressure Code, American Society of Mechanical Engineers, 1971.

- 19.3-22 Letter from E. J. Mroczka to NRC, Response to NRC Bulletin No. 88-08, Thermal Stresses in Piping Connected to Reactor Coolant System, September 20, 1988.
- 19.3-23 Letter from D. H. Jaffe to E. J. Mroczka, NRC Bulletin 88- 08, Thermal Stresses in Piping Connected to Reactor Coolant Systems (TAC No.69636, 69651 and 69653), September 25, 1991.
- 19.3-24 Letter from E. J. Mroczka to NRC, NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification, February 28, 1989.
- 19.3-25 Letter from J. F. Stolz to E. J. Mroczka, NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification - Evaluation of Westinghouse Owners Group Bounding Analysis (TAC No. 72136 and 72145), August 6, 1990.
- 19.3-26 Letter from J.F. Stolz to E. J. Mroczka, Pressurizer Surge Line Thermal Stratification, Bulletin 88-11, Millstone Unit 3 and Haddam Neck (TAC No. 72145 and 72136), July 31, 1991.
- 19.3-27 Letter from J. F. Opeka to NRC, NRC Bulletin 88-11 - Pressurizer Surge Line Thermal Stratification Final Submittal of Plant-Specific Reports, May 1, 1992.
- 19.3-28 Letter from J. F. Stolz to J. F. Opeka, Response to NRC Bulletin No. 88-11 - Pressurizer Surge Line Thermal Stratification for Haddam Neck Plant (TAC No. M72136) and Millstone 3 (TAC No. M72145), July 9, 1992.
- 19.3-29 ANSI B31.1, Power Piping Code, American Society of Mechanical Engineers, 1967.
- 19.3-30 NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, U.S. Nuclear Regulatory Commission.
- 19.3-31 NUREG/CR-6583, Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels, U.S. Nuclear Regulatory Commission.
- 19.3-32 NUREG/CR-5704, Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steel, U.S. Nuclear Regulatory Commission.
- 19.3-33 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.
- 19.3-34 Regulatory Guide 1.89, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.
- 19.3-35 Regulatory Guide 1.97, Instrumentation of Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, U. S. Nuclear Regulatory Commission.

19.3-36 WCAP-14535A, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination, Westinghouse Energy Systems, November 1996.

19.4 TLAA SUPPORT PROGRAMS

19.4.1 ELECTRICAL EQUIPMENT QUALIFICATION

Program Description

The Electrical Equipment Qualification program corresponds to the Time-Limited Aging Analysis (TLAA) support program described in NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.” The program applies to certain electrical components that are important to safety and could be exposed to post-accident environmental conditions, as defined in 10 CFR 50.49. The EEQ program ensures the continued qualification of this equipment during and following design basis accidents. The program determines the necessity for, and frequency of, component replacement or refurbishment in order to maintain the qualification of the equipment. Performance of preventive maintenance and surveillance activities, and monitoring of normal ambient conditions, ensure that components remain within the bounds of their original qualification and provide a basis for extending qualified life through re-analysis.

The acceptance criterion is that the equipment remains within the bounds of its qualified life such that after maximum normal service conditions, the equipment retains sufficient capacity to perform its required safety function during design basis accident conditions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.4.2 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary program mitigates fatigue cracking caused by cyclic strains in metal components of the reactor coolant pressure boundary. This is accomplished by monitoring and tracking the number of critical thermal and pressure transients for selected Reactor Coolant System components to ensure that the number of design transient cycles is not exceeded during the plant operating life.

The acceptance criterion is the fatigue usage factors bounded by the design usage factors. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

19.5 EXEMPTIONS

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analyses as defined in 10 CFR 54.3 have been identified.

19.6 LICENSE RENEWAL COMMITMENTS

Table 19.6 - 1, License Renewal Commitments, provides a listing of the license renewal commitments.

19.6.1 REFERENCES FOR SECTION 19.6

19.6-37 Letter from Leslie N. Hartz to NRC, Millstone Power Station Units 2 and 3, Response to Request for Additional Information License Renewal Applications, August 13, 2004 (Serial No.: 04-398).

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS

Item	Commitment	Source	Schedule ^a
1	The existing inspection program will be modified to include those battery racks that require monitoring for license renewal, but are not already included in the program.	Battery Rack Inspections	Prior to Period of Extended Operation
2	Implementing procedures will be modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks.	Battery Rack Inspections	Prior to Period of Extended Operation
3	A baseline inspection of the in-scope buried piping located in a damp soil environment will be performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Prior to Period of Extended Operation
4	The maintenance and work control procedures will be revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures will include the inspection for the loss of material due to selective leaching which will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Prior to Period of Extended Operation
5	The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be established.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Prior to Period of Extended Operation

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
6	Fuse holders meeting the requirements will be evaluated prior to the period of extended operation for possible aging effects requiring management. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The program (if needed for fuse holders) will consider the aging stressors for the metallic clips.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Complete.
7	Procedures will be developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Prior to Period of Extended Operation Not to Exceed a 10 year Frequency Thereafter
8	A baseline visual inspection will be performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation.	Fire Protection Program	Prior to Period of Extended Operation
9	Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years will be included in the Fire Protection Program.	Fire Protection Program	Prior to The Sprinkler Heads Achieving 50 Years of Service Life Not to Exceed a 10 Year Interval Thereafter
10	The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns will be enhanced to provide expectations that identify the requirements for the inspection of aging effects.	General Condition Monitoring	Prior to Period of Extended Operation

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
11	<p>In-scope cable found to be submerged will be subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation will also address the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test will use a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables, or other appropriate testing. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program.</p>	<p>Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements</p>	<p>Prior to Period of Extended Operation During the Corresponding 10 Year Interval (If Applicable)</p>
12	<p>The Infrequently Accessed Areas Inspection Program will be established.</p>	<p>Infrequently Accessed Areas Inspection Program</p>	<p>Prior to Period of Extended Operation</p>
13	<p>Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC and IASCC), and loss of pre-load for baffle and former-assembly bolts and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, will be submitted to the NRC for approval.</p>	<p>Inservice Inspection Program: Reactor Vessel Internals</p>	<p>At Least Two Years Prior to Period of Extended Operation</p>

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
14	Augmented inspection of the Millstone Unit 3 core barrel holddown spring will be performed. In particular, the inspection will detect gross indication of loss of preload as an aging effect. As an alternative to performing an augmented inspection, the holddown spring will be replaced.	Inservice Inspection Program: Reactor Vessel Internals	At Least Two Years Prior to the Period of Extended Operation
15	Millstone will follow the industry efforts investigating the aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description will be submitted at least two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 50.54.21(a)(3).	Inservice Inspection Program: Systems, Components and Supports	At Least Two Years Prior to Period of Extended Operation
		Steam Generator Structural Integrity	
16	The existing inspection program will be modified to include those lifting devices that require monitoring for license renewal, but are not already included in the program.	Inspection Activities: Load Handling Cranes and Devices	Prior to Period of Extended Operation
17	Implementing procedures and documentation will be modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal in Commitment 16.	Inspection Activities: Load Handling Cranes and Devices	Prior to Period of Extended Operation
18	The implementing procedures will be modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program.	Structures Monitoring Program	Prior to Period of Extended Operation
19	The Structures Monitoring Program and implementing procedures will be modified to include all in-scope structures.	Structures Monitoring Program	Prior to Period of Extended Operation

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
20	Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade.	Structures Monitoring Program	Prior to Period of Extended Operation
21	The Structures Monitoring Program and implementing procedures will be modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged.	Structures Monitoring Program	Prior to Period of Extended Operation
22	The maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason.	Structures Monitoring Program	Prior to Period of Extended Operation
23	Appropriate inspections of sealants and caulking used for moisture intrusion prevention in and around aboveground tanks will be performed.	Tank Inspection Program	Prior to Period of Extended Operation
24	Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, will be performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities.	Tank Inspection Program	Prior to Period of Extended Operation A frequency consistent with scheduled tank internals inspection activities
25	The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and will be included on the respective Tank Inspection Program inspection plan.	Tank Inspection Program	Complete
26	Changes will be made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities.	Work Control Process	Prior to Period of Extended Operation

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
27	Consistent with 10 CFR 54.21(c)(1),(iii), the effects of environmentally assisted fatigue for those specific locations with a CUF greater than 1.0 will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary program. If the specific locations are not repaired, replaced, or successfully re-analyzed, a modified inspection program description, including a comparison to the 10 program elements of NUREG-1801 program, will submitted to the NRC for approval.	Environmentally Assisted Fatigue TLAA	Prior to Period of Extended Operation
28	For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally-embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3.	Inservice Inspection Program: Systems, Components and Supports	Prior to Period of Extended Operation
29	Millstone will follow industry efforts that will provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will also implement the appropriate recommendations resulting from this guidance. Until these recommendations are available, Millstone 3 commits to using the pressurizer surge line nozzle as a leading indicator to address environmental effects on fatigue of pressurizer sub-components during the period of extended operation.	Environmentally Assisted Fatigue TLAA	Prior to Period of Extended Operation

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
30	A baseline visual inspection will be performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one:	Closed-Cycle Cooling Water System	Prior to Period of Extended Operation
	Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger,		
	Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and		
	Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.		
31	Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching will be performed on a representative sample of locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation.	Work Control Process	Prior to Period of Extended Operation
32	A review of the Work Control Process inspection opportunities for each material and environment group, supplemental to the initial review conducted during the development of the LRA, will be performed. Baseline inspections will be performed for the material and environment combinations that have not been inspected as part of the Work Control Process.	Work Control Process	Prior to Period of Extended Operation
33	Calibration results for cables tested in situ will be reviewed to detect severe aging degradation of the cable insulation. The initial review will be completed prior to entering the period of extended operation and will include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Prior to Period of Extended Operation Not to Exceed a 10 Year Frequency Thereafter

TABLE 19.6 - 1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule ^a
34	The in scope cables in Unit 3 duct lines # 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and # 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C and 35D) will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	“Prior to period of extended operation.” Complete. Subsequent testing will not exceed a 10 year frequency.
35	In addition to the testing specified in Commitment 34, a representative sample of in-scope medium voltage cables will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Prior to Period of Extended Operation Not to Exceed a 10 Year Frequency Thereafter
36	Millstone Unit 3 will complete the SAMA evaluation of the ability to manually control the Turbine Driven Auxiliary Feedwater Pump. If this SAMA is cost beneficial (i.e., can be accomplished without a hardware modification), a Severe Accident Management Guideline (SAMG) addressing this mitigation strategy will be developed.	Severe Accident Mitigation Alternatives (SAMA) Analysis (Reference 19.6-37)	Prior to Period of Extended Operation
37	The pressurizer spray head assembly will be either replaced or inspected utilizing the best currently available (at the time of inspection) techniques for detecting cracking resulting from SCC.	Inservice Inspection Program: Systems, Components and Supports	Prior to Period of Extended Operation

a. The period of extended operation is the period of 20 years beyond the expiration date of the Unit’s previous 40 year Operating License.